U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-410/87-27

Docket No. 50-410

License No. NPF-69

Licensee: Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Facility Name: Nine Mile Point Nuclear Station, Unit 2

Inspection At: Scriba, New York

Inspection Conducted: August 3-12, 1987

Inspectors: L. J. Wink — 8/20/87
L. J. Wink, Reactor Engineer — date

L. J. Wink FOR — 8/20/87
D. J. Florek, Lead Reactor Engineer — date

Approved by: P. W. Eselgroth — 8/20/87
P. W. Eselgroth, Chief, Test Programs Section, DRS — date

Inspection Summary: Inspection on August 3-12, 1987 (Report No. 50-410/87-27)

Areas Inspected: Routine, unannounced inspection by two region-based inspectors of the overall power ascension test program including test witnessing and test results evaluation, QA interfaces, and independent measurements and verifications.

Results: No violations were identified.

NOTE: For acronyms not defined, refer to NUREG-0544, "Handbook of Acronyms and Initialisms."

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DETAILS

1.0 Persons Contacted

Niagara Mohawk Power Corporation

*R. Abbott, Station Superintendent
A. Anderson, Station Shift Supervisor
R. Bodily, Shift Test Supervisor
G. Carlisle, Lead STD&A Engineer
J. Conway, Power Ascension Manager
W. Davey, Station Shift Supervisor
P. Eddy, Site Representative, New York State, PSC
R. Gayne, Assistant Superintendent of Operations
*D. Helms, Lead Shift Test Supervisor
M. Jones, Superintendent of Operations
G. Moyer, Station Shift Supervisor
H. Pao, Shift Test Supervisor
B. Rudd, Shift Test Supervisor
*W. Wambsgan, Assistant Superintendent of Operations
*L. Wolf, Site Licensing Engineer

NRC Personnel

*W. Cook, Senior Resident Inspector
C. Marschall, Resident Inspector
N. Perry, Reactor Engineer
W. Schmidt, Resident Inspector

*Denotes those present at the exit meeting on August 12, 1987.

The inspector also contacted other members of the licensee's Operations, Technical, Test and QA staffs.

2.0 Power Ascension Test Program (PATP)

2.1 References

- Regulatory Guide 1.68, Revision 2, August 1978, "Initial Test Program for Water Cooled Nuclear Power Plants."
- ANSI N18.7-1976, "Administrative Controls and Quality Assurance for Operations Phase of Nuclear Power Plants."
- Nine Mile Point Unit 2 (NMP-2) Technical Specifications, July 2, 1987.
- Nine Mile Point Unit 2 Final Safety Analysis Report (FSAR) Chapter 14, "Initial Test Program."

- Nine Mile Point Unit 2 Safety Evaluation Report.
- Nine Mile Point Unit 2 AP-1.4, Startup Test Phase, Revision 3

2.2 Overall Power Ascension Test Program

The inspector held discussions with the Power Ascension Manager (PAM), the Lead Startup, Design and Analysis (STD&A) Engineer and other members of the PATP staff to assess the status of testing, the test results evaluation process and the preparation and approval of test procedures. In addition, the inspector attended the daily power ascension management meetings and Site Operations Review Committee (SORC) meetings involving the PATP.

At the beginning of the inspection period, the unit was at 16% of rated thermal power and tuning was in progress on the Electro-Hydraulic Control (EHC) system. The unit had been shutdown on July 26, 1987, in compliance with technical specifications, when service water intake temperature exceeded 77°F. A restart occurred on July 27, 1987, and on July 29, 1987, the unit exceeded 5% of rated thermal power for the first time following the interim resolution of problems with the Offgas system.

On August 6, 1987, the main turbine was rolled to rated speed (1800 RPM) for the first time. Following turbine testing and generator checks, the generator was synchronized to the grid on August 8, 1987 and loaded to approximately 100 MWe. Testing planned for Test Condition 1 was substantially completed during the inspection period. On August 9, 1987, the licensee conducted the remote shutdown demonstration (see discussion of N2-SUT-28-1, Shutdown from Outside the Main Control Room, in paragraph 2.3) and entered a planned ten day outage.

Major work activities planned for the outage include modification of feedwater piping supports (temperature stratification), removal of condensing pots on main steam line flow instrumentation and modifications to the Offgas and RWCU systems to correct identified problems. Following completion of outage activities, the licensee plans to restart and commence testing in Test Condition 2.

2.3 Power Ascension Test Witnessing

Scope

The inspector witnessed the performance of the power ascension tests discussed below. The performance of this test was witnessed to verify the attributes previously defined in Inspection Report No. 50-410/86-64, Section 2.3.

Discussion

N2-SUT-14-1, RCIC System

This test was performed on August 5, 1987, at a reactor pressure of 950 psig and 17% of rated thermal power. This was the second of two required demonstrations of the ability of RCIC to automatically start and inject into the reactor pressure vessel from a cold, standby condition (minimum time required was 72 hours since last system operation). The inspector observed the conduct of the pre-test briefing and assignment of responsibilities. The automatic start sequence was begun by arming and depressing the system initiation pushbutton. The inspector observed overall system performance to be excellent and independently confirmed that RCIC achieved rated flow to the reactor pressure vessel within the required 30 seconds. During the course of the test the inspector observed excellent coordination of operations and test personnel and the overall proficient conduct of the test. The inspector also observed operations personnel monitoring suppression pool temperature as required by technical specifications during testing which adds heat to the pool.

N2-SUT-26-1, Relief Valve Testing

This test was performed on August 4, 1987, at a reactor pressure of 951 psig and 18% of rated thermal power. During this test, data was also gathered on the acoustic monitors and the response of the relief valve discharge piping.

The inspector attended the initial pre-test briefing. Since the test extended over two shifts, the inspector also observed the shift turnover process and the pre-test briefing of the oncoming shift. During the performance of the test, the inspector monitored diverse plant parameters to confirm expected response and verify positive indication of steam flow through each relief valve and positive indication of valve closure following testing. All relief valves performed as expected. However, during the testing of PSV-126, the acoustic monitor failed to give the proper indication on valve closure. The failure was considered a Level 1 test exception and the inspector observed administrative compliance with the requirements for handling these exceptions. The Station Shift Supervisor and Shift Test Coordinator informed the Station Superintendent and Power Ascension Manager of the exception and placed testing in a hold condition. A proposed resolution was formulated (reduce the acoustic monitor's gain and re-test) and forwarded to SORC for review. Following the SORC review and approval by the General Superintendent, the valve was successfully re-tested. The overall coordination and conduct of the test was excellent. During the test, the inspector also verified compliance with technical specifications surveillance requirements for suppression pool temperature monitoring and vacuum breaker operability testing.

N2-SUT-28-1, Shutdown from Outside the Main Control Room

This test was performed on August 9, 1987. The initial conditions for the demonstration were established as 17.8% rated thermal power, reactor pressure 950 psig, core flow 38.8 mlb/hr with the main turbine generator on-line and supplying a load of 115 MWe. The test was also witnessed by the resident inspector (see Inspection Report No. 50-410/87-20).

The inspector witnessed the conduct of the test and a test briefing for the cooldown demonstration. The test consisted of two phases, a hot shutdown demonstration which required scramming the plant from outside the control room, closing the MSIVs and, utilizing the remote shutdown panel, stabilizing the plant conditions and assuring the ability to remove decay heat. The test crew then transferred control to the main control room to establish conditions for the cooldown phase which involved establishing the residual heat removal system in the shutdown cooling mode and further cooldown of the plant.

At 12:36 p.m. the licensee simulated a control room evacuation and the test crew (1 SRO - 3 RO) assumed their positions. The licensee continuously maintained an additional shift crew in the control room during this test. At 12:41 the reactor was locally scrammed. The MSIVs and main feed pumps were also locally tripped. The test crew maintained pressure control by use of the relief valves which were opened 6 times. The test crew initiated RCIC and demonstrated injection ability and secured RCIC due to sufficient inventory. The test crew also demonstrated suppression pool cooling. This portion of the test was concluded at 1:30 p.m. Control was then transferred back to the main control room.

At 6:00 p.m. the licensee began the cooldown demonstration (prior to the resumption of testing the licensee conducted a test briefing). The test crew warmed the lines for RHR from the remote shutdown panel and initiated RHR in the shutdown cooling mode with an initial reactor coolant temperature of 310°F. When the reactor coolant temperature reached 250°F the test was terminated and control returned to the main control room at 8:10 p.m.

During the test the licensee experienced a few minor problems but they did not hamper the performance of the test. The licensee was recording these to be resolved after the test was completed.

Findings

No unacceptable conditions were identified.

2.4 Power Ascension Test Results Evaluation

Scope

The power ascension test results discussed below were evaluated for the attributes identified in Inspection Report 50-410/86-54, Section 2.1. The test results evaluated were in the process of being technically reviewed by the power ascension test program staff and had not received a SORC review nor been accepted by the General Superintendent. The inspector will verify the formal review and management acceptance of these test results during a future, routine inspection of the power ascension test program.

A summary of significant test results and identified test results deficiencies is provided in the following discussion.

Discussion

N2-SUT-11-1, LPRM Flux Response

There were no acceptance criteria associated with this test. It was performed to verify that the LPRM detectors and their associated electronics were properly connected. The flux response of the LPRM detectors was monitored via control room indication and the process computer (OD-8 edit) while an adjacent control rod was manipulated. All LPRM detectors responded properly during the test.

N2-SUT-12-1, APRM Calibration

The APRMs were calibrated conservatively by means of a manual heat balance (N2-RCPCP-1, Core Thermal Power Heat Balance, Revision 0). The core thermal power was determined to be 584.8MWt (17.6% of rated) and all APRMs were adjusted to read 18.0%. The technical specification setpoints for scrams and rod blocks were verified. All acceptance criteria were satisfied.

N2-SUT-14-1, RCIC System

The RCIC system was tested six times in various configurations and at various steam supply pressures to span the design operating range of the system. Testing consisted of actual vessel injections at reactor pressures of 150 psig and rated, including both hot and cold quick starts, a surveillance demonstration with flow returned to the CST and a demonstration of system operability from the Remote Shutdown panel. Time to rated flow was measured at reactor pressures of 150 psig and rated during both hot and cold quick starts. All times measured were well within the acceptance criterion limit of 30 seconds. Level 2 test exceptions were identified for minor steam leaks on the governor valve and turbine trip-throttle valve at rated

pressure. During the surveillance demonstration to the CST a Level 2 test exception occurred when the peak turbine speed (measure - 4882 RPM) exceeded the limit of 4777 RPM (overspeed trip avoidance margin). This exception is attributable to difficulties in adjusting a throttle valve in the test return line to the CST to simulate appropriate reactor pressure and is not indicative of a system performance problem. The throttle valve must be adjusted to simulate a discharge pressure equal to reactor pressure plus at least 100 psi. The actual setting of the valve is difficult to adjust precisely and, in this case, was positioned in such a way as to simulate reactor pressure plus approximately 450 psi (a condition which could not occur during actual operation) which resulted in a high turbine speed being required to achieve rated flow. Results of actual vessel injections have demonstrated acceptable overspeed trip avoidance margin. All other test acceptance criteria were satisfied during these tests.

N2-SUT-19-1, Core Performance

This initial verification of the core thermal-hydraulic limits was performed at 17.0% of rated thermal power (565 MWt) and 35.5% of rated core flow (38.55 MLb/hr). Core power was calculated using a manual heat balance and the power distribution was evaluated by the BUCLE program. All acceptance criteria were satisfied and the results are summarized below:

Parameter	Measured Value	Limit
LHGR (kW/ft)	3.34	$\leq$ 13.40
CPR	4.929	$\geq$ 1.55
APLHGR (kW/ft)	2.89	$\leq$ 12.00

N2-SUT-22-1, Pressure Regulator

This test was performed at 13.8% of rated thermal power to demonstrate adequate transient response of the pressure regulator with controlling pressure via the main turbine bypass valves. All acceptance criteria were satisfied.

N2-SUT-23-1, Feedwater System

This test was performed at approximately 16% of rated thermal power to demonstrate adequate transient response of the feedwater high pressure, low flow control valves (FWS-LV55A/B). All acceptance criteria were satisfied.

N2-SUT-26-1, Relief Valve Testing

The functional test of the relief valves was performed at 17% of rated thermal power and a reactor pressure of 952 psig. Two test exceptions were identified. The first, involving a Level 1 acceptance criterion failure, occurred when the acoustic monitor indicating light for relief valve PSV-126 did not properly indicate the closed position of the valve following its manual opening and subsequent closure. The acoustic monitor's gain was reduced and the valve was satisfactorily recycled to close this exception. The second, a Level 2 acceptance criterion failure, was observed following the completion of relief valve cycling when two relief valve tailpipe temperatures did not return to within 10°F of their initial readings, indicating minor valve "weeping". The inspector will follow the resolution of this deficiency during a future routine inspection. All other acceptance criteria were satisfied.

N2-SUT-28-1, Shutdown from Outside the Main Control Room

This test was reviewed to verify that the minor problems encountered during the shutdown demonstration were properly documented in the results package and that plans were made to resolve them. The inspector found the test documentation complete and clearly organized. All problems encountered were documented and required corrective actions identified. The inspector will review the final resolutions to these identified problems during a future routine inspection. The acceptance criterion for this test was satisfied.

N2-SUT-29-1, Recirculation Flow Control

This test was performed at 17% of rated thermal power to demonstrate adequate transient response of the recirculation flow control system while operating in the position (Loop Manual) control mode. All acceptance criteria were satisfied and the maximum rate of change of valve position was demonstrated to be less than the technical specification limit of 11%/sec.

N2-SUT-33-1, Drywell Piping Vibration

All acceptance criteria for steady state vibration of the reactor recirculation system piping were satisfied.

N2-SUT-71-HU, Residual Heat Removal System

This test was performed to close plateau exception N2-PP-HU-3 which allowed the test to be deferred from Test Condition Heatup due to an insufficient temperature difference between the suppression pool and service water. The test demonstrated the ability of the RHR system to operate in the Suppression Pool Cooling mode with adequate heat

exchange capacity (design - 41.6 MBtu/hr). The measured heat exchange capacity was 84.9 MBtu/hr for the "A" loop and 95.3 MBtu/hr for the "B" loop.

N2-PP-1, Test Plateau 1

The inspector review the test plateau procedure to insure that all testing planned for Test Condition 1 had been accomplished and that all test exceptions identified had been properly documented and resolved. The review determined that portions of four tests involving the steam condensing mode of the RHR system had not been completed due to identified steam leakage in the system. The inspector determined that the licensee plans to correct these leaks during the current outage and to perform these tests prior to exceeding 25% of rated thermal power during the ascent to Test Condition 2 power-to-flow conditions. Additional testing also remained to complete the confirmation of proper IRM/APRM overlap. This testing will also be performed during the startup to Test Condition 2. The review of outstanding test exceptions revealed no problems that would require resolution prior to beginning Test Condition 2. The inspector concluded that the licensee's plan to complete required testing and resolve open test exceptions was adequate.

Findings

No unacceptable conditions were identified.

3.0 QA Interface with the PATP

During the witnessing of the power ascension test discussed in paragraph 2.3, the inspector observed QA engineer's performing surveillances of the testing activities.

No unacceptable conditions were noted.

4.0 Independent Measurements and Verifications

The inspector independently verified conformance with the acceptance criteria for RCIC system time to rated flow during the witnessing of power ascension test N2-SUT-14-1, RCIC system - Test Condition 1, and for positive indication of steam discharge during the manual actuation of relief valves during the witnessing of power ascension test N2-SUT-26-1, Relief Valve Testing TC-1, as discussed in paragraph 2.3. In addition, during the evaluation of the results of power ascension test N2-SUT-26-1, Relief Valve Testing TC-1, as discussed in paragraph 2.4, the inspector independently calculated the steam flow through each relief valve, using GETARS traces of bypass valve response, and verified that no major

blockages existed in the discharge paths of the valves. In all cases, the inspector's measurements and verifications agreed with those of the licensee.

No unacceptable conditions were noted.

5.0 Exit Interview

At the conclusion of the inspection on August 12, 1987, an exit meeting was held with licensee personnel (identified in Section 1.0) to discuss the inspection scope, findings and observations as detailed in this report. At no time during the inspection was written materials provided to the licensee by the inspector. Based on the NRC Region I review of this report and discussions held with licensee representatives during the inspection, it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions.

<u>Attachment A</u>

<u>POWER ASCENSION TEST RESULTS EVALUATED</u>

N2-SUT-11-1	LPRM Flux Response, Revision 2, completed August 1, 1987
N2-SUT-12-1	APRM Calibration - Test Condition 1, Revision 1, completed August 1, 1987
N2-SUT-14-1	RCIC System - Test Condition 1, Revision 2, completed August 8, 1987
N2-SUT-19-1	Core Performance - Test Condition 1, Revision 2, completed August 5, 1987
N2-SUT-22-1	Pressure Regulator - Test Condition 1, Revision 1, completed August 4, 1987
N2-SUT-23-1	Feedwater System, Revision 1, completed August 3, 1987
N2-SUT-26-1	Relief Valve Testing TC-1, Revision 2, completed August 5, 1987
N2-SUT-28-1	Shutdown from Outside the Main Control Room, Revision 1, completed August 9, 1987
N2-SUT-29-1	Recirculation Flow Control - TC1, Revision 2, completed August 5, 1987
N2-SUT-33-1	Drywell Piping Vibration - Test Condition 1, Revision 1, completed August 5, 1987
N2-SUT-71-HU	Residual Heat Removal System, Revision 1, completed August 6, 1987